**NRC Report**
**August 2010**

1. Amendment to 10 CFR 50.55a – ASME Code Edition/Addenda

A proposed rule was published in the Federal Register on May 4, 2010 (75 FR 24324). The public comment period closed on July 19, 2010, and the NRC received 19 public comment letters.

Under the current plan, the final rule would be published in May 2011.

The proposed amendment to 10 CFR 50.55a would incorporate by reference:

- The 2005 Addenda through 2008 Addenda of Section III, Division 1, and Section XI, Division 1, of the *Boiler and Pressure Vessel Code*;

- The 2005 Addenda and 2006 Addenda of the *Code for Operation and Maintenance of Nuclear Power Plants*;

- Code Case N-722-1, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials Section XI, Division 1;" and

- Code Case N-770, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1."

The proposed rule would also:

- clarify which portions of Section III are approved for use by applicants and licensees
- identify which portions of Section III are NRC requirements, and which portions of Section III are not required to be implemented by 10 CFR 50.55a
- substitute the word "condition(s)" for the words "limitation(s)" "modification(s)" and "provision(s)" throughout 50.55a for consistency
- clarify the time frame for licensees to submit requests for relief based on impracticality for IST and ISI
- allow the use of 1994 Edition of NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," when using the 2006 Addenda of Section III of the ASME B&PV Code and later editions and addenda
- remove conditions throughout 50.55a that are no longer necessary and renumbering paragraphs as appropriate

Finally, the NRC also requested comments on what the scope of the ASME B&PV Code edition and addenda rulemaking should be; how often the NRC should incorporate Code editions and addenda into 10 CFR 50.55a; and in what ways the NRC should communicate the scope, schedule for publishing the rulemakings in the Federal Register, and status of the 10 CFR 50.55a rulemakings to external users.

2. ASME Code Case Rulemaking/Regulatory Guides

On June 2, 2009, Draft Revision 35 to RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” draft Revision 16 to RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,”and draft Revision 3 to RG 1.193 “ASME Code Cases Not Approved for Use,” were published in the Federal Register (74 FR 26303) for public comment. The guides address Code Cases from Supplement 2 to the 2004 Edition through Supplement 0 to the 2007 Edition (Supplement 0, 2007 Edition also serves as Supplement 12 to the 2004 Edition). The public comment period closed on August 17, 2009. The draft final guides (including responses to public comments) have been reviewed and approved by the program offices, and the Office of the General Counsel (OGC) provided a “No Legal Objection” to final publication on July 21, 2010. RG 1.193 was published on July XX, 2010. The final rule, RG 1.84, and RG 1.147 will be published in September 2010.

It was discussed in the NRC Report to the ASME in May 2010 that OGC had advised the staff that the federal courts were being stricter relative to re-noticing, i.e., all actions being considered for adoption in the final rule, including the NRC’s basis for doing so, must be provided in the proposed rule absent very special circumstances. The staff has had further discussion with OGC regarding re-noticing. The NRC usually receives a number of comments each rulemaking cycle requesting approval of a later version of a Code Case, i.e., Code Case N-508-3 is listed in the draft guide and it is suggested that Code Case N-508-4 be listed in the final guide. The inclusion of later versions of Code Cases in a final guide will be determined on a case-by-case basis; however in general, a later version can only be listed if it addresses an administrative/editorial issue or clarifies the intent of the Code Case. Later versions deemed to include new or revised technical provisions need to be noticed for public comment.

The NRC staff has completed its review of Supplements 1 − 11 to the 2007 Edition. Draft Revision 36 to RG 1.84, draft Revision 17 to RG 1.147, draft Revision 2 to RG 1.192, and draft Revision 4 to RG 1.193 have been reviewed by the cognizant NRC offices. The draft guides will address Supplements 1 – 9 to the 2007 Edition. The goal is to publish these guides for public comment shortly after Revision 35 to RG 1.84 and Revision 16 to RG 1.47 have been published as final guides.

The staff is considering addressing the issues raised by Raymond A. West in a petition for rulemaking dated December 14, 2007, and revised on December 19, 2007, in the proposed rulemaking for Revision 17 of Regulatory Guide 1.147.

3. Risk-Informed Activities

On December 1, 2009, Westinghouse submitted a revised Pressurized Water Reactor Owners Group (PWROG) plan to the NRC for implementation of WCAP- 16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (Agencywide Documents Accession and Management System [ADAMS] No. ML093370133). The implementation plan was revised as a result of three recent changes in inspection requirements: 1) MRP-139 and ASME Section XI Code Case N-770 - Inspection and Mitigation of Alloy 82/182 Reactor Vessel Nozzle Welds, 2) MRP-227 - Inspection and Evaluation Guidelines for PWR Reactor Vessel Internals, and 3) 10 CFR 50.61a- Alternate Pressurized Thermal Shock Rule.

On June 28, 2010, (ML101750602) the NRC issued a License Amendment for the Shearon Harris Nuclear Power Plant that transitions the existing fire protection program to a risk-informed, performance-based program based on National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, in accordance with Title 10 of the *Code of Federal Regulations,* Paragraph 50.48(c). NFPA 805 allows the use of performance-based methods, such as fire modeling and fire risk evaluations, to demonstrate compliance with the nuclear safety performance criteria.

4. Generic Activities on Material Degradation/PWR Alloy 600/182/82 PWSCC

In 2006 ASME started the development of a Code Case for inspection of Alloy 82/182 butt welds. Code Case N-770 was developed to address inspection of these welds, and the NRC included the Code Case in the recently published proposed amendment to 10 CFR 50.55a (see #1 above). The NRC staff previously provided comments relative to the proposed conditions on the Code Case to the cognizant ASME committees.

The NRC staff continues to monitor and evaluate operating experience to ensure that the current inspection schedules are adequate.

The staff developed Regulatory Issue Summary 2010-07, “Regulatory Requirements for Application of Weld Overlays and Other Mitigation Techniques in Piping Systems Approved for Leak-Before-Break,” (see #11 below) on the regulatory requirements for application of weld overlays and other mitigation techniques in piping systems approved for leak-before-break (LBB).

5. New Reactor Licensing Activities

The New Reactor Licensing public web-site [http://nrr10.nrc.gov/NRO/new-rx-status/index.cfm] has a list of expected new nuclear power plant applications, and an estimated schedule by fiscal year for new reactor licensing applications.

*New Reactor Licensing Status*

As of July 15, 2010, the status of new reactors licensing under 10 CFR Part 52 is as follows:

*Design Certification*

NRC has issued four design certifications to date (ABWR, System 80+, AP-600, and AP-1000). These are certified in 10 CFR Part 52, Appendices A, B, C, and D, respectively. The NRC is currently reviewing four design certifications:

- General Electric-Hitachi’s ESBWR (first passive BWR)
- AREVA’s EPR (evolutionary pressurized-water reactor)
- Mitsubishi Heavy Industries’ US-APWR (advanced pressurized water reactor)
- AP-1000 Revision 17 (first amended design certification)

*Early Site Permits (ESPs)*

NRC has issued four ESPs to date (Clinton, Grand Gulf, North Anna, and Vogtle). The NRC's issuance of the Vogtle ESP on August 26, 2009, is the first to be based on a specific technology (AP-1000) and the first to include a limited-work authorization (LWA). The NRC received an application for an ESP for the Victoria County Station submitted by Exelon on March 25, 2010. This is the first ESP application for a greenfield site with no specific technology established at this time.

The NRC received an ESP application for the PSEG site in New Jersey (same site as Hope Creek and Salem 1&2). The ESP application was tendered on May 25, 2010, and is currently undergoing an acceptance review. The NRC's acceptance decision is expected at the end of July 2010.

*Combined License (COL) Applications*

The North Anna, Unit 3 COLA was revised on June 29, 2010, to change its standard plant design from an ESBWR to a US-APWR.

NRC is currently reviewing 17 COL applications (27 new reactor units):

- 1 ABWR — South Texas Project 3 and 4
- 7 AP-1000 — Vogtle 3&4, William S. Lee Station 1&2, Shearon Harris 2&3, V.C. Summer 2&3, Levy County 1&2, Bellefonte 3&4, and Turkey Point 6&7
- 4 ESBWR — Fermi 3, Grand Gulf 3*, River Bend 3*, Victoria County 1 and 2*
- 3 EPR — Calvert Cliffs 3, Nine Mile Point 3*, Bell Bend
- 2 US-APWR — Comanche Peak Units 3 and 4, North Anna 3

* NRC staff review suspended at request of applicant.

*Advanced Reactors Program*

NRC has established an advanced reactors program in the Office of New Reactors. Currently there are no applications under review, but several applications are expected to be submitted in the next three years including:

- High Temperature Gas-Cooled Reactors:
  - Next Generation Nuclear Plant (DOE) – Design Certification application expected as early as FY 2013
- Small and Medium-size LWRs:
  - NuScale – Design Certification application expected early FY 2012
  - mPower (B&W) – Design Certification application expected late FY 2012
  - IRIS (Westinghouse) – Design Certification application expected as early as Q3 2013

- o Liquid Metal Reactors (LMR)
- o Toshiba 4S – Design Approval application expected as early as FY 2012
- o GE PRISM application for prototype as early as 2012

*NRO Vendor Inspection*

The NRO vendor inspection program is described in Inspection Manual Chapter (IMC) 2507, "Construction Inspection Program, Vendor Inspection." This IMC will be implemented by various Inspection Procedures (IPs) including:

IP 43002: Routine Inspections of Nuclear Vendors;
IP 43003: Reactive Inspections of Nuclear Vendors;
IP 43004: Inspection of Commercial-Grade Dedication Programs;
IP 43005: NRC Oversight of Third Party Organizations Implementing Quality Assurance Requirements; and
IP 36100: Inspection of 10 CFR Parts 21 and 50.55(e) Programs for Reporting Defects and Noncompliance.

*FY 10 Vendor Inspection Plans*

- Commercial grade dedication organizations
- Manufacturing for valves (all new reactor Design Centers)
- Forgings suppliers for AP-1000, EPR
- Manufacturing for steam generator tubes EPR and AP-1000
- STP ABWR reactor vessel fabrication in Japan
- STP ABWR mechanical component fabrication in Japan
- AP-1000 Modular Construction Facilities

*Vendor Inspection Reports completed, issued and planned inspections*

- Shaw Power Group, Charlotte, NC, March 2010 – issued
- Sulzer Pumps, March, 2010, Chattanooga TN – issued
- South Texas Project Units 3&4 (ABWR Design Certification amendment) Bay City, Texas - issued
- Westinghouse (Seismic Structural Code), Cranberry Woods, PA, May, 2010 – issued
- Westinghouse (Shield Building Testing at Purdue), W. Lafayette, IN, May 2010 - issued
- Sandvik Materials Technology, Sandviken, Sweden (Areva EPR SG tubes), June 2010 – issued
- International Quality Consultants, Butler, PA, June 2010 – issued
- Mangiarotti, Udine, Italy (AP-1000 components) – completed
- Black and Veatch, Kansas City, MO – scheduled
- IHI, Yokohama, Japan (AP-1000 containments & STP ABWR), scheduled

Vendor Inspections continue to identify findings related to commercial grade dedication activities and inadequate Part 21 programs for evaluating and reporting of defects that could cause a substantial safety hazard.

Previously issued NRC inspection and trip reports can be located at

http://www.nrc.gov/reactors/new-reactors/oversight/quality-assurance/vendor-insp.html

*Multinational Design Evaluation Programme (MDEP) Codes and Standards Working Group*

MDEP is a multinational initiative to develop innovative approaches to leverage the resources and knowledge of mature, experienced national regulatory authorities who will be tasked with the regulatory design review of new reactor plant designs. One of the issue-specific working groups established under the MDEP organization is the Codes and Standards Working Group (CSWG) whose goal is to achieve harmonization of Code requirements for pressure-boundary components.

Harmonizing pressure-boundary Codes used by member countries would ensure a consistent level of quality and safety in the design of pressure-boundary components such as the reactor vessel, piping, pumps, and valves and allow components manufactured in other countries to be used in member countries with a relatively minor review and reconciliation of Code differences. Such an approach would significantly simplify the licensing of nuclear power plants and reduce the burden on the regulatory authorities on an international scale.

The MDEP/CSWG has been working with standards development organizations (SDOs) from several countries (i.e., U.S., Japan, Korea, France, Canada and, recently, the Russian Federation) for the past 2 years to compare each countries' pressure-boundary Code requirements for Class 1 vessels, piping, pumps and valves to the requirements of the ASME Boiler and Pressure Vessel Code, Section III. Similarities and differences are being documented in a database table. The Code-comparison effort is the first step to achieve harmonization of pressure-boundary codes and standards. The Code-comparisons are essentially complete for Class 1 vessels, piping, pumps and valves for Korea, Japan, and France with Canada following shortly thereafter. Russia recently initiated a comparison of its pressure-boundary code for Class 1 vessels, piping, pumps and valves. The SDOs from Japan, France, Canada, Russia and the U.S. met at the OECD Nuclear Energy Agency's offices on April 8-9, 2010 to discuss with the MDEP/CSWG their results and significant findings from its Code comparisons. The MDEP/CSWG presented to the SDOs its conceptual plan to harmonize pressure-boundary codes and standards on an international level. The MDEP/CSWG plans to issue a letter to each of the SDOs requesting support to harmonize pressure-boundary codes and standards and to consider how further divergence of code requirements can be prevented. The next MDEP/CSWG meeting is tentatively planned to be held in Vancouver, Canada, in November 2010 in conjunction with the ASME Boiler Code meeting.

Multinational Design Evaluation Program (MDEP) Vendor Inspection Cooperation Working Group (VICWG) activities:

NRC staff continues involvement for international cooperation of vendor oversight through the MDEP and through interactions with other international regulatory bodies. The staff has met with the Japan Nuclear Energy Safety Organization (JNES), the Japan Nuclear and Industrial Safety Agency (NISA), the French Nuclear Safety Authority (ASN), the Korean Institute of Nuclear Safety (KINS), the Chinese regulator (NNSA) and the regulator from Great Britain.

The MDEP VICWG members continue to allow opportunities for NRC staff participation and observation of vendor inspections conducted by regulatory authorities from other countries and for opportunities where participation and observation of NRC vendor inspections by representatives of regulatory authorities from other countries is possible. VICWG objectives include: explore international regulators' vendor oversight requirements and programs; apply lessons learned; exchange vendor inspection insights; and identify areas where international cooperation can yield tangible benefits.

On April 19-23, 2010, NRO staff observed the British Regulator perform an engineering procurement inspection of AREVA design activities in Paris, France.

On May 10-12, 2010, NRO staff participated at the meeting of the VICWG in Paris, France. Discussion included lessons learned on MDEP vendor inspections conducted to date, a general discussion of vendor inspection activities, and a discussion on common quality assurance criteria for member counties.

NRC Regulatory Guide (RG) 1.28 Revision 4 Update

In October 2007, ASME requested NRC Endorsement of the NQA-1-2008 Edition. NRC Draft Guide DG-1215 for proposed Revision 4 of Regulatory Guide 1.28 was posted on the NRC website for public comment in July 2009 and includes endorsement of NQA-1-2008 edition and 2009-1a addenda. The NRC has completed its review of 33 public comments received from stakeholders.

Revision 4 to RG 1.28 was issued on June 7, 2010. NRC endorsement of NQA-1-2008 Edition and NQA-1a-2009 Addenda includes Part I and Part II requirements and identifies specific regulatory positions.

RG 1.28, Revision 4 can be located at the NRC website at:

http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/rg/

2nd NRC Workshop on Vendor Oversight for New Reactor Construction

The NRC held its 2nd Workshop on Vendor Oversight for New Reactor Construction on June 17, 2010, in New Orleans, LA, to share insights and lessons learned for companies supplying components and services for new reactor construction.

The purpose of this workshop was to bring together NRC staff, regulated utilities, vendors of nuclear components, and other interested stakeholders to discuss recent issues. This one-day NRC workshop was held upon completion of the periodic Nuclear Procurement Issues Committee (NUPIC) meeting to maximize industry and vendor participation. Vendor inspection topics at this workshop included both the NRC's and industry's perspectives on vendor oversight for new reactors; the ASME nuclear survey process; the NRC enforcement policy as it applies to vendors; counterfeit, fraudulent, and suspect items; safety culture; and vendor perspectives on third party inspections, audits, and surveys. In addition to presentations by the NRC staff, there were presentations by NUPIC, NEI, EPRI, ASME, and two nuclear vendors.

The workshop had over 550 people in attendance from 13 countries, 45 international and domestic utilities (and applicants), 10 regulators/government agencies, three (3) industry groups (i.e., EPRI, NIAC, and NUPIC), and 203 vendors/suppliers.

Additional workshop information is available on the NRC Web site here: http://www.nrc.gov/reactors/new-reactors/oversight/quality-assurance/vendor-oversight.html.

## 6. LICENSE RENEWAL ACTIVITIES

There are several on-going activities in license renewal. The current status of applications and approvals is:

*Current status of applications, staff reviews and approvals is:*

- 59 units approved (Beaver Valley on November 5 and Susquehanna on November 24)
- 13 applications (19 units) under review
    - 2 (2 units) awaiting final approval (Pilgrim and Vermont Yankee)
    - 2 (4 units) completed ACRS full committee (Indian Point 2 & 3, Prairie Island 1 & 2)
    - 0 (0 units) completed ACRS subcommittee
    - 7 (10 units) awaiting ACRS subcommittee (Cooper 5/10, Duane Arnold 6/10, Crystal River 6/10, Kewaunee 7/10, Palo Verde 9/10, Hope Creek 11/10 and Salem 12/10)
    - 2 (3 units) applications received (Diablo Canyon 1 & 2 and Columbia)
- 6 applications with scheduled application dates through 2011
    - April-June 2010 – Seabrook
    - August 2010 – Davis-Besse
    - October-December 2010 – South Texas Project 1 & 2
    - July 2011 – Grand Gulf
    - September 2011 – Limerick 1 & 2
    - October 2011 – Callaway
    - Others staggered out to 2017

Four plants have entered the operating period beyond 40 years:

- Oyster Creek – April 9, 2009
- Nine Mile Point Unit 1 – August 22, 2009
- Ginna – September 19, 2009
- Dresden Unit 2 – December 22, 2009

Upcoming plants in 2010 are H.B. Robinson, Monticello, and Point Beach Unit 1.

*Revision of Generic Aging Lessons Learned (GALL) Report (NUREG-1801)*

NRC has an on-going internal activity to develop an update of the GALL report and the License Renewal Standard Review Plan (SRP). This revision is comprehensive in nature, including consideration of aging management programs (AMPs), aging management review (AMR) line items from the GALL tables, and the SRP. Sources of information for the proposed revisions are:

- Interim Staff Guidance documents
- Comments from the industry (Nuclear Energy Institute)
- Plant operating experience (generic communications, etc.)
- Lessons learned and precedents from LRA reviews
- The NRR RES Proactive Materials Degradation Assessment (PMDA).

NRC staff will modify the GALL Report to address concerns raised by ASME about use of Section XI Code Editions, Relief Requests, and Code Cases for license renewal, consistent with the summary of the NRC-ASME teleconference held on August 10, 2009 (see (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092440512, available on the NRC web site [http://www.nrc.gov]).

Schedule:

- December 2009 – draft portions of documents were made available on the web
- May 12, 2010 – Notice of Availability for public comment [ML101320132]: Draft NUREG-1800, Revision 2: “A Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants;” and Draft NUREG-1801, Revision 2; “A Generic Aging Lessons Learned (GALL) Report;” and announcement of public workshop.
- December 2010 – final revised GALL and SRP to be issued

Status and schedule can be tracked at:

http://www.nrc.gov/reactors/operating/licensing/renewal/guidance/updated-guidance.html

*Technical Issues*

Recent reviews and plant operating experience have identified issues in the following areas:

- Neutron Absorbers
  - Information Notice 2009-26, “Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool,” issued on October 28.
  - Draft Interim Staff Guidance (ISG) for Boral and other neutron absorber materials was issued for public comment on December 1 (LR-ISG-2009-01, “Staff Guidance Regarding Plant-Specific Aging Management Review and Aging Management Program for Neutron-Absorbing Material In Spent Fuel Pools”). The final ISG will be issued soon.

- Buried Piping

  - Recent operating experience, including tritium releases. NRC has initiated on-going interactions with NEI, EPRI, INPO and NACE.
- Socket Welds
  - Consideration of the need for non-visual examinations to ensure integrity of these welds.
  - Industry reviewing/evaluating operating experience
- Metal Fatigue
  - Additional information routinely requested for NRC reviews (dissolved oxygen, cycle counting, etc.). RIS 2008-030 describes the need to use six stress components instead of one to assure conservative fatigue calculations. The Office of Nuclear Regulatory Research is considering additional work in the area of environmental fatigue and has initiated discussions with EPRI.
- Containment Liner
  - Corrosion identified at several plants.
  - An item was introduced on the agenda of Section XI, Subsection IWL to assess the need to identify early detection methods for containment liner plate degradation/corrosion. Discussion of this issue is continuing in working group meetings.
  - NRC has initiated an activity to review operating experience and assess likelihood of corrosion occurrence.
- Concrete Containment
  - Delamination at tendon thickness location identified at one plant. Conditions not identified at a similar plant.
- Medium Voltage Cables
  - Cables in submerged environment not qualified for continuous submergence.

7. Buried Piping

Recent leaks from buried piping at nuclear power plants have caused the NRC to undertake a focused look at how underground piping is designed, maintained, and inspected to ensure structural integrity and to prevent leaks that could harm the environment. These leaks generated significant stakeholder interest, including inquiries from several congressmen. On December 2, 2009, the NRC staff responded to the Chairman's memorandum dated September 3, 2009, ADAMS No. ML092460648, tasking the staff to describe the activities currently underway or planned addressing the issue of leaks from buried piping. The response is SECY-09-0174, "Staff Progress in Evaluation of Buried Piping at Nuclear Reactor Facilities," and can be found at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2009/.

A new Code Case N-XXX is under development to address underground piping systems. The staff has identified several areas that are not yet addressed, or require significant additional detail. The staff has provided its comments to Section III.

With regard to the integrity of buried piping systems and the prevention of groundwater contamination, the staff has initiated discussions with cognizant ASME committees regarding three issues. The first issue is that the Section III Code Cases currently under development do not address the ASME Code requirement for the design and arrangement of system components to allow for adequate access and clearances to conduct examination and tests. Lack of access for the inspection of buried safety-related Class 3 service water system (SWS) piping has been identified as a serious issue relative to ensuring piping integrity. The second issue is the need to re-examine the scope of buried piping in the ASME Code. Presently, the ASME Code only addresses safety-related Class 3 SWS piping. Leakage of contaminated fluids from other buried piping systems has been reported. Leakage from these systems can affect system operation and may have radiological impacts. Accordingly, the staff submitted an Issue Sheet to the Subgroup on Industry Experience for New Plants for consideration. The staff proposed that Section III consider the development of requirements addressing accessibility for inspection of buried piping. Systems to be included would be based on considerations such as function and consequence. The staff also proposed that Section XI consider the development of requirements addressing the inspection of buried piping for new plants. The inspection of buried piping at operating plants is the third issue. The staff is concerned that: ASME Code required testing and surveillance requirements for Class 3 buried piping do not appear to be sufficient to identify corrosion, degradation and leakage; and leaks from other buried piping systems carrying tritium were discovered through voluntary licensee monitoring for radioactive tritium in groundwater monitoring wells rather than through inspection, testing, or monitoring of the piping.

With regard to Class 3 buried piping, the Class 3 buried piping pressure boundary has degraded and become compromised at several plants. In some cases, the degradation was significant but had not yet challenged structural integrity. Current ASME Code requirements are not sufficient to identify either degradation or a leak. If left undiscovered, degradation of buried Class 3 piping could progress to a point that structural integrity is threatened, particularly for piping that experiences general coating failure followed by general corrosion. If such a system contains tritium, groundwater monitoring would indicate the presence of a leak. For those piping systems that do not contain tritium, however, neither groundwater monitoring nor inspection or testing will identify degradation, and the piping will continue to deteriorate until there is loss of function.

With regard to buried piping other than Class 3, industry assessments have shown that it is important to maintain the integrity of these systems as they can affect the reliability of plant operation and can have radiological and environmental impacts. The industry has initiated a number of activities. Section XI should assess these activities with regard to the need for inspection and testing.

With regard to Code Case N-755 regarding the use of high density polyethylene piping for underground systems, the staff has identified issues to Section III related to design life, joining, and non-destructive examination that will need to be addressed for the staff to endorse the Code Case.

8. Information Notice 2010-12: Containment Liner Corrosion

On June 18, 2010, the NRC issued Information Notice (IN) 2010-12, “Containment Liner Corrosion”, to inform licensees of recent issues involving containment corrosion. The NRC expects recipients to review the information for applicability to their facilities and to consider actions, as appropriate, to avoid similar problems. The IN was prompted by events at the Beaver Valley Power Station, Unit 1, Brunswick Steam Electric Plant, Unit 1, and Salem Nuclear Generating Station Unit 2.

On April 23, 2009, during a Subsection IWE visual examination, blistered paint was identified on the liner at the Beaver Valley Power Station, Unit 1. Follow-up cleaning activity revealed a rectangular area of approximately 1 inch (horizontal) x 3/8 inch (vertical) that penetrated through the entire liner plate thickness. Ultrasonic testing (UT) of the surrounding area showed liner thinning within an area of approximately 10 square inches. Removal of the corroded section of liner revealed a partially decomposed piece of wood approximately 2 inches x 4 inches x 6 inches embedded in the concrete behind the section of the liner. The wood was left behind as a result of inadequate housekeeping and quality assurance practices during the original construction of the containment wall in the early 1970s. Additional information is available in Beaver Valley Licensee Event Report 50-334/2009-003-00, dated June 18, 2009, and Beaver Valley Power Station, Unit 1, NRC Routine Inspection Report 05000334/2009006, dated July 6, 2009, which can be found on the NRC's public Web site under Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML091740056 and ML091870328, respectively.

During a refueling outage in 2008 at Brunswick Steam Electric Plant, Unit 1, the licensee performed a VT-1 visual inspection of the primary containment penetration sleeve for the personnel air lock and found two bulged areas. The discovery of thinned areas on the bulges led the licensee to perform UT examinations of the entire Unit 1 personnel air lock penetration sleeve. These additional UT inspections identified many discrete locations that were below the minimum wall thickness established by the design-basis containment liner specification. During construction, the outside diameter of the sleeve was wrapped with two layers of 1/4 inch felt and the felt was covered with a layer of 60 mil ethylene propylene film. The felt was intended to permit the sleeve to expand when subjected to thermal loading. The licensee's evaluation determined that the bulges were caused by corrosion product buildup between the sleeve and the concrete backing. This corrosion was caused by the felt that wrapped around the outside of the containment penetration sleeve; which became wet during the original construction.

In October 2009, at Salem Nuclear Generating Station Unit 2, the licensee inspected the containment moisture barrier (the silicone RTV [room temperature vulcanizing] seal between the concrete floor and containment liner) and found heavy corrosion on the containment liner within 6 inches of the concrete floor. This area of the containment liner was considered inaccessible because it was normally covered by an insulation package that consisted of a layer of sheet metal, a layer of plastic sheeting, and a layer of insulation. The licensee had not inspected the containment liner areas covered by this insulation because ASME Code Section XI allowed an exemption for inaccessible areas. In response to this discovery, and as a conservative approach to the license renewal process, the licensee decided to enhance inspections of the containment liner above the moisture barrier within about 6 inches of the concrete floor and to randomly inspect several other areas that were covered by the insulation package. To perform the inspections, the licensee removed that portion of the insulation package that extended below the lower leak detection channel for the entire containment liner circumference, and cut through and removed the insulation package for four other randomly selected areas. Licensee inspections in these four areas identified some corrosion but subsequent ultrasonic measurements did not indicate significant wall loss. The licensee determined that the source of the moisture that caused the liner corrosion at the joint between the containment liner and concrete floor was service water leakage from the containment fan coil units and associated piping. It was determined that previous containment liner inspections were not performed adequately.

The NRC has begun an assessment to better understand the possible mechanisms responsible for through-wall corrosion of containment liners. The NRC staff has also engaged Section XI committee members to devise a formal tracking mechanism to monitor industry experience and events involving containment liner corrosion. Subsection IWE could then be updated using insights from these events.

9. Information Notice 2010-08: Welding and Nondestructive Examination Issues

On April 9, 2010, the NRC issued IN 2010-08, "Welding and Nondestructive Examination Issues," to alert licensees of recent operating experience involving welding and nondestructive examination. On October 1, 2008, at Prairie Island Nuclear Generating Plant, NRC inspectors reviewed the weld control records of the licensee contractor that performed machine gas tungsten arc welding of the structural weld overlay on a pressurizer surge nozzle dissimilar metal weld. The NRC inspectors noted that the documented weld speed on several weld control records had not changed as would be expected and as required by procedure. To control the weld heat input (an essential welding variable), it is necessary to control the weld head travel speed. The operating instruction specified that welders would use the values for piping/nozzle radius and the desired travel speed to determine the appropriate travel speed setting for the welding machine. In this case, contrary to welding procedure specifications, the welders did not determine and enter a new speed into the welding machine as the piping/nozzle radius increased as weld layers were added. Notwithstanding, a subsequent engineering evaluation determined that based on the travel speeds recorded in the weld control record, the resulting heat inputs were within the acceptable range. While evaluating this weld speed issue, the welding contractor discovered that incorrect welding parameters had been used to apply the first layer of the temper bead weld over a section of the ferritic nozzle adjacent to a stainless steel butter interface. Specifically, the welders had failed to change temper bead welding parameters when transitioning from the butter to the nozzle, a requirement which was not delineated in the operating instruction. The resulting heat input on the ferritic nozzle exceeded that allowed by the welding procedure specifications. As a result, the weld had to be removed (ground out) and rewelded. Additional information is available in Prairie Island Nuclear Generating Plant, Units 1 and 2, NRC Integrated Inspection Report (IR) 05000282/2008005; 05000306/2008005, dated February 10, 2009, and can be found on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML090420033.

In May 2009, the NRC inspectors identified welding issues involving the construction of the Louisiana Energy Services, National Enrichment Facility (LES NEF). The following are three examples of failure to implement American Society of Mechanical Engineers (ASME) B31.3 "Process Piping" requirements for welding and nondestructive examination: (1) Progressive radiography sampling was not proceduralized to comply with the requirements of ASME B31.3. The NRC inspectors reviewed weld records and identified examples where a designated lot of random radiography samples was not expanded when results revealed a weld defect for work performed by a welder/welding operator in Category M Fluid Service; (2) Welders were not qualified in accordance with ASME Section IX for manual tack welding of pipes, as required by ASME B31.3; and (3) Weld reinforcement height exceeding the maximum allowed by ASME B31.3 for circumferential butt welds on pipe. Additionally, NRC inspectors identified examples where LES NEF did not meet requirements regarding commercial grade dedication of cascade component welds. Additional information regarding the above examples is available in NRC IR 70-3103/2009-002, dated June 26, 2009, at ADAMS Accession No. ML091770643 and NRC IR 70-3103/2009-007, dated January 27, 2010, at ADAMS Accession No. ML100271177.

On October 22, 2008, inspectors for the French regulatory authority, Authorité de Sûreté Nucléaire, inspected the welding records of Società delle Fucine (SdF), a subcontractor under AREVA who in turn is the primary contractor for the construction of Flamanville-3. The inspectors reviewed documentation and quality assurance measures applied to the weld test samples to support the manufacturing of an intermediate collar ring for a pressurizer. The inspectors found that the subcontractor used welding rods that were not qualified and did not conform to the American Society for Testing and Materials (ASTM) E208, the International Organization for Standardization (ISO) 9001, and the subcontractor's own procedures. The welding rods were considered unqualified because they were not tested before performing drop-weight testing. The inspectors found that the subcontractor SdF was aware that unqualified weld rods were used for the test samples and decided to not generate a nonconformance report, which is contrary to ASTM E208, ISO 9001, and the subcontractor's own procedures. The subcontractor only generated a nonconformance report following unsuccessful tests of the samples taken from the intermediate collar ring of the pressurizer. The French regulators determined that without the necessary documentation to demonstrate conformance with the French nuclear construction code, it did not have the assurance that the subcontractor manufactured the pressurizer components in accordance with specified requirements. Also noted was that the primary contractor (AREVA) had not prevented the nonconformances by its subcontractor.

NUREG-1425, "Welding and Nondestructive Examination Issues at Seabrook Nuclear Station: An Independent Review Team Report," dated July 28, 1990, describes lessons learned regarding licensee radiographic and welding programs. (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML090300351). These occurrences illustrate the importance of licensee oversight of contractors or subcontractors, including direct observation of welding activities while in progress when possible, to ensure welders adhere to welding procedure specifications and QA requirements; specifically, 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

10. Information Notice 2010-07: Welding Defects in Replacement Steam Generators

On April 5, 2010, the NRC issues IN 2010-07, "Welding Defects in Replacement Steam Generators," to inform licensees of welding defects that were associated with the manufacturing of replacement steam generators (RSGs). Southern California Edison (SCE) contracted Mitsubishi Heavy Industries (MHI) to manufacture four RSGs in Japan for installation at San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. During a routine visual inspection by MHI after completion of the ASME Section III primary and secondary side hydrostatic pressure test on the SONGS Unit 3 "B" RSG, a 5-inch long surface flaw (crack) was discovered in the dissimilar metal weld between the divider plate (Alloy 690) and the channel head (low-alloy steel (LAS)). The flaw formed between the LAS and the Alloy 152 butter. Further inspection revealed additional defects, including separation of almost all of the Alloy 152 butter under the divider plate and some stainless steel cladding adjacent to the divider plate from the LAS substrate in certain locations. MHI determined that the separation in the SONGS Unit 3

RSGs followed the fusion line between the Alloy 152 butter/stainless steel cladding and the LAS substrate. The weld joint was prepared by removing the stainless steel cladding from the RSG surface using air carbon-arc gouging (ACAG) and surface grinding to prepare for the deposition of Alloy 152 as a butter pass. The root cause of the separation was associated with the ACAG technique. The ACAG resulted in higher carbon content and areas of higher hardness in the vicinity of the fusion line between the butter pass and the LAS substrate. During subsequent surface preparation by grinding, MHI did not ensure that all of the surface carbonized material was removed. The regions of higher hardness and variations in surface conditions led to unfavorable metallurgical properties at the interface between the Alloy 152 butter and LAS substrate.

The welding defects identified at the fabrication facility on the RSGs for SONGS Unit 3 are unlike the weldability issues that are typically observed in the welding of nickel-based alloys. The fabricator followed approved welding procedures for dissimilar metal welding of Alloy 690 to LAS and had recently built two RSGs for SONGS Unit 2 according to these procedures without problems. However, for the SONGS Unit 3 RSGs, the fabricator requested and the licensee approved a deviation to allow using an alternative method (in this case ACAG) to prepare the LAS surface for butter application. According to the ASME Code, Section IX, this deviation did not require the requalification of the welding procedure because this aspect of the weld joint preparation was not considered an essential variable. ACAG is not specifically covered in Section III of the ASME Code; however, ASME Code, Section XI, IWA-4461 covers the qualification and use of a thermal removal process like ACAG. In addition, 10 CFR 50.55a(b)(2)(xxiii) states: "The use of provisions to eliminate the mechanical processing of thermally cut surfaces in IWA-4461.4.2 of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of 10 CFR 50.55a are prohibited." Although all specific requirements or standards were met, this event illustrates that control over all aspects of welding ASME Code Class 1, 2, and 3 components can prevent welding defects like those found in the RSGs for SONGS Unit 3 from occurring.

## 11. Regulatory Issue Summary 2010-07, Leak-Before-Break Systems

On June 8, 2010, the NRC issued Regulatory Issue Summary 2010-07, "Regulatory Requirements for Application of Weld Overlays and Other Mitigation Techniques in Piping Systems Approved for Leak-Before-Break," to remind addressees of the regulatory requirements for application of weld overlays and other mitigation techniques in piping systems approved by the NRC for leak-before-break (LBB). LBB analyses are performed to demonstrate that the probability of fluid system rupture is extremely low. Weld overlays and other mitigation techniques are being used to mitigate Alloy 82/182 butt welds against primary water stress-corrosion cracking (PWSCC) in PWRs. A weld overlay may change the weld geometry of the original weld upon which the LBB analysis was based, thus potentially invalidating the original LBB analysis.

The governing requirement for LBB is General Design Criterion (GDC) 4. GDC 4 requires that structures, systems, and components be designed to accommodate the environmental and dynamic effects of postulated pipe ruptures. The NRC staff previously approved plant-specific LBB analyses for the reactor coolant system (RCS) piping at all PWR facilities and approved plant-specific LBB analyses for some RCS branch piping at a limited number of PWRs. The

NRC staff approved these LBB analyses under GDC 4 using the guidance in NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 3, "Evaluation of Potential for Pipe Breaks," issued November 1984 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093170485), or SRP Section 3.6.3, Revision 0. When the NRC approved LBB analyses in the 1980s and 1990s for the currently operating fleet of PWRs, RCS butt welds had not exhibited PWSCC, and, therefore, the NRC staff concluded that PWR RCS piping was not susceptible to cracking failure from the effects of corrosion. Since 2000, PWSCC has occurred in the RCS systems of a number of PWRs. RIS 2008-25, "Regulatory Approach for Primary Water Stress-Corrosion Cracking of Dissimilar Metal Butt Welds in Pressurized-Water Reactor Primary Coolant System Piping," dated October 22, 2008, discusses PWSCC in Alloy 82/182 RCS piping butt welds. In RIS 2008-25, the NRC staff discussed the actions taken to address the potential effects of PWSCC.

12. NRC Reactor Pressure Vessel Embrittlement Workshop

## INVITATION TO PARTICIPATE

***DEVELOPMENT OF PREDICTIVE MODELS OF NEUTRON IRRADIATION EMBRITTLEMENT IN REACTOR PRESSURE VESSEL STEELS TO SUPPORT WORLDWIDE EFFORTS ON NUCLEAR POWER PLANT LIFE EXTENSION***

**Kick-Off Meeting**
**September 14-15, 2010**

**Rockville, Maryland (USA)**

**SPONSORED and ORGANIZED by**

**U. S. Nuclear Regulatory Commission and the Oak Ridge National Laboratory**

## THE PROPOSED EFFORT

It is now well established that neutron embrittlement of RPV steels is a complex phenomenon. The magnitude of embrittlement depends on the interplay of a number of environmental (fluence, flux, temperature, etc.) and compositional (copper, nickel, manganese, phosphorus, silicon, etc.) variables. Considerable data exists regarding the effects of neutron embrittlement on both mechanical properties (e.g., strength, hardness, impact energy, fracture toughness) and on microstructural properties. However, due to the expense of working on irradiated materials, individual data sets tend to be limited and rarely include complete information on both mechanical and microstructural properties for the same material exposed in both test and power reactors. Both of these factors (i.e., the complexity of the phenomena that creates neutron irradiation damage and the lack of comprehensive data sets to quantify its effects) have inhibited progress toward the development and validation of a comprehensive physically-based model that is sufficiently robust to enable confident prediction of future embrittlement trends. The aim of this project is to develop and validate such a model using the following process:

1. **Establish what is now known** by assembling, in a single place, a database to contain the most comprehensive collection possible of world-wide knowledge concerning the effects of neutron irradiation embrittlement on the mechanical and microstructural properties of RPV steels.
2. Use this database to **establish interim models** based on the assembled information, and to **identify areas where current knowledge is deficient**.
3. **Fill gaps in existing knowledge** by performing the necessary additional experiments and by developing the necessary theoretical frameworks.
4. **Develop final models** based on already existing information, plus the information developed in this project.
5. **Update the models** as new information from surveillance and test reactor irradiations becomes available.

It is recognized that a project of such scope cannot be conducted by any one organization. Indeed, to be successful, such a project requires the participation of organizations and individuals representing a wide range of interests (e.g., regulators, industry, universities, research organizations, and national laboratories) worldwide.

## KICK-OFF MEETING

You and your organization are invited to participate in a two day kick-off meeting that will be held on September 14-15, 2010, in Rockville, Maryland (USA). **The objectives of the meeting are to establish a common appreciation among the participants of the current state of knowledge and the gaps therein, and to identify a series of next steps that can be taken to develop a participation agreement for data sharing among interested attendees.** It is anticipated that the meeting agenda, which will be developed in further detail in the near future, will be structured generally as follows:

- **September 14, 2010**
  - To establish a common starting point, presentations will be made concerning regulatory needs, industrial needs, currently available mechanical property data, currently available microstructural data, and the current understanding of the physical processes underlying the radiation damage of RPV steels.
  - Nationally and internationally, there are several on-going efforts to both assemble existing irradiation data, and to collect more data. Representatives of these efforts will provide summaries of their efforts to provide meeting attendees with an appreciation of the current state of on-going activity
  - After lunch, the attendees will form a committee-of-the-whole to discuss the following four topics:
    - ✓ Protocols for data sharing
    - ✓ Structure of a storage database, or databases
    - ✓ Mechanical property data – what should be retained?
    - ✓ Microstructural data – what should be retained?

    Two discussion leaders will facilitate each discussion. After the general meeting adjourns, the discussion leaders will caucus to compare notes and chart a course of action.

- **September 15, 2010**
  - The day will begin with a closed meeting of the discussion leaders, who will develop a proposal for moving forward.
  - At approximately 11A.M. the general meeting will convene, with a presentation to the committee-of-the-whole of the proposal made by the discussion leaders.
  - This proposal will be discussed by the committee-of-the-whole to elicit comments and suggested improvements.
  - The meeting will conclude with a list of actions placed upon the discussion leaders. The aim of these actions will be to enable the drafting of a participation agreement. The participation agreement will be finalized within one month after the meeting, and then circulated to the attendees who have expressed interest in continued participation in the project. It is anticipated that technical progress toward the aims of the project will be achieved based on in-kind contributions received from all participants.

## LOCATION/ACCOMMODATIONS

The Workshop will be held at:

**The Legacy Hotel and Meeting Centre**
**1775 Rockville Pike**
**Rockville, Maryland 20852**
**Tel. (301) 881-2300**
**www.thelegacyrockville.com**

The Legacy is on the Washington DC METRO's RED line and can be easily accessed by Metro directly from Reagan National Airport (DCA). A block of rooms has been allocated for attendees to the Workshop. For the evenings of September 13-14, 2010, **attendees should book directly with the hotel BEFORE August 14, 2010, to receive the special Workshop rate** of $229/night**,** plus tax ($10 additional person). (**Note:** The Legacy Hotel has the workshop identified as **Group # 357083**. Please refer to this Group number when making reservations to receive the special Workshop rate.)

## CONTACTS FOR PARTICIPANTS

Meeting participants who are interested in making short presentations on relevant topics should contact the **technical coordinator**, who is listed below. Organizations and individuals interested in attending the meeting, and those who need information on local details, should contact the **administrative coordinator**, who is also listed below.

| | |
|---|---|
| **Technical Coordinator:** | **Mark Kirk, U.S. NRC**<br>Mark.Kirk@nrc.gov |
| **Administrative Coordinator:** | **Angie Scott, ORNL**<br>scottar@ornl.gov<br>**Tel. (865) 241-0331** |